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Prospects of VVER-SKD reactor in a closed fuel cycle

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Abstract

At the new centure's begin eight countries with developed nuclear power industry took part under the aegis of the IAEA in research of innovative nuclear reactors and fuel cycles to choose a nuclear power system with fast reactors based on a closed fuel cycle (CFC) and to perform joint R&D in this direction. An agreement was reached on the use of based on proven technologies CNFC-FR (Closed Nuclear Fuel Cycles and Fast Reactors), as a reference system for common assessment.

Common principles, however, did not eliminate among participating countries essential discrepancies neither in existing nuclear power systems nor in development strategies, which has led to discrepancies in implementation of CFC. Gas and lead coolant are proposed along with sodium, and nitride (more dense) as well as metallic fuel – along with MOX, so the different fuel cycles.

Since 2000, IV-generation reactors cooled with water at supercritical state (SCWR – Supercritical Water-Cooled Reactors) are developed in many countries. Construction of demonstration facilities are planned to 2025, followed by commercial nuclear power systems. Development of SCWR will correct the development of nuclear power industry strategy and the CFC in several countries.

This paper considers characteristics of CFC implementation in Russia, milestones, dates, problems arising. The use of fast neutron spectrum SCWR reactors within CFC is justified.

Keywords: International forum Gen-IV; Fast reactor; Sodium; Lead; Supercritical water; MOX fuel; Closed fuel cycle; Spent nuclear fuel; Transmutation of minor actinides.

Introduction

In January 2000, the "International forum Generation-IV" (GIF) was started by the USA Department of Energy (DOE), with the aim to initiate and steer R&D on nuclear power facilities of fourth generation, by identifying potential fields of international cooperation [1].

Main directions of R&D for gen-IV reactors are defined by the goal of GIF, which is to provide:

- sustainable development to satisfy society's energy demand without damaging the natural environment, by ecologically rational generation of energy and long-lasting nuclear fuel together with decrease of nuclear waste amount;
- safety and reliability of nuclear power facilities, guarantying exceptionally low probability and degree of core damage;
- economical competitiveness of nuclear power facilities, because of beneficial lifecycle cost as compared to another energy sources, and due to level of financial risk, comparable to other energy projects;
- nonproliferation of nuclear weapon and nuclear weapon materials, together with improvement of physical protection against terrorism [2].

The assessment has been made by a group of 100 experts – leading specialists in nuclear power engineering – resulted in a choice of six base concepts of gen-IV reactors, to be developed in the framework of GIF. In the present work only three of them, developed to the most extent, are considered: sodium-cooled reactors (SFR), lead-cooled (LFR) and reactors with supercritical water (SCWR).

Another research was conducted (2005–2007) under the direction of IAEA, the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), with participation of eight

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countries with developed nuclear power. The task was to define a nuclear power system based on a closed fuel cycle (CFC) with fast-spectrum reactors, milestones and dates of its implementation, and to identify fields of cooperate R&D. The participating countries agreed to apply as a reference for comparison the "reference system" CNFC-FR (Closed Nuclear Fuel Cycles and Fast Reactors), ready for deployment in next 20–30 years and based on proven technologies of sodium coolant, pellet MOX fuel and improved technology of water treatment [3].

Both above mentioned programs aimed to identify promising directions of nuclear power future development, taking into account the necessity to close the fuel cycle and organization of international cooperation for solving the above mentioned problems.

One of the promising gen-IV reactors identified in the GIF program, is the supercritical water (SCW) cooled reactor, SCWR. Conceptual designs of SCWR are being elaborated by more than 45 organizations in 16 countries with developed nuclear power.

Starting 2000, the international symposium on supercritical water-cooled reactors is organized every two years, with about 100 contributions (presentations). The most recent, ISSCWR-7 was held in Helsinki, on 15–18 March 2015.

The SCWR concept is based on the once through coolant scheme, with SCW as coolant. Implementation of such reactor will increase the thermal efficiency factor up to 44–45%, increase fuel breeding ratio, decrease metal intensity and amount of construction, and enhance ecology.

The countries participating in SCWR development within GIF consider the development of a reactor with thermal neutron spectrum based on existing experience with PWR and BWR, as a top-priority task. On the next stages, after technology has been settled, transition to fast-spectrum reactor is assumed.

A thermal-spectrum SCWR is characterized by significantly improved plant economy, requires however enriched uranium that is followed by increased amount of spent fuel and minor actinides. In general, such a reactor will not satisfy one of the far goals – the closing of fuel cycle.

Development of SCWR raises several scientific and technical problems to solve, such as:

- Development and verification of neutronics, hydrodynamics and thermal physics computational codes specific to SCW,for fuel assembly (FA) and whole reactor core.
- Design of fuel pins and assemblies and justification of its operational availability.
- Analysis of reactor stability at normal operation and during accident transients.
- Choice of high-temperature resistant materials for fuel pins, featuring high corrosion- and crack resistance.
- Justification and development of optimal water chemistry regime.

Some problems are investigated at stand and loop experimental facilities, however, to solve the bunch of problems as a whole, to justify the SCWR technology and for later licensing, an experimental test facility is needed.

In the GIF roadmap, the main effort for the next 10 years targets the development of a small-power experimental reactor. The completion of the concept design is planned for the first 5 years, while for the next 5 years – detailed design and construction of the facility.

Present work describes characteristics of nuclear power development in Russia and prospects of gen-IV reactors applied to CFC, milestones, dates of implementation and problems arising. The SCWR development prospects and their use in CFC-based nuclear power systems are justified.

Characteristics of nuclear power development and closure of fuel cycle in Russia

An increase of nuclear energy fraction is planned from 16% (23 GWel) to 25% (80 GWel) to 2050, in order to decrease fossil fuel consumption.

Possible programs with commissioning of 1,2 GWel/year (one BN-1200 per year) are considered, together withconstruction and commissioning of the BREST lead-cooled reactors.

Peculiar is the concept of on-site fuel cycle facility, including non-aqueous treatment of MOX, nitride and metal fuel including vibrocompaction.

The goals of such a program is to develop the nuclear power system that includes sodium- and lead-cooled NPPs with fast neutron spectra, nuclear fuel recycling (reprocessing) and refabricating facilities, removal of radioactive wastes from the technological cycle, that meets the following requirements:

- Elimination of accidents requiring evacuation and relocation of inhabitants.
- Closure of nuclear fuel cycle to full utilization of uranium energy potential.
- Technological foundation of nonproliferation (subsequent rejection on uranium enrichment in nuclear power industry, weapon Pu breeding in blankets and isolation during radwaste handling, shortening of nuclear material transportation).
- Reaching a balance between radioactivity of disposed radwaste and mined uranium.
- Fast reactor NPPs capital cost reduction (at least to the level of fossil-fueled power plants) due to technological and design solutions inherent to fast neutron reactors only.

The following implementation plan of major components of this system is assumed [4,5].

The BN-800 reactor is commissioned in 2014, supplemented to 2017 with the 1st stage of the on-site nuclear fuel cycle facility for MOX, and to 2020 – for nitride fuel. The BREST-OD-300 detailed project design shall be ready to 2016, and it shall be constructed, together with its on-site fuel cycle facility for nitride fuel, to 2020. Completion of the whole platform, including BN-1200, is planned to 2025.

Major parts of the fuel cycle shall be located on two sites: at the Beloyarsk NPP with BN-600 and BN-800 reactors, where the BN-1200 reactor and its on-site fuel cycle facility is planned, and in Seversk on the site of "Siberian Chemical Combine" (SKhK), where BREST-OD-300 and its on-site fuel cycle facilities shall be constructed.

Development of such complex system of CFC, consisting of two different types of fast reactors cooled with sodium and lead with different type of fuel (oxide, nitride, metallic and carbide), reveals many problems.

Technical problems

While BN reactor technology is being developed during about 60 years and the most recent BN-800 and BN-1200 can be considered as a “standard”, the lead-cooled reactors have no equal in the world and the most experience in similar facilities, cooled by lead-bismuth eutectic has been gained by Russia.

Large experience in design and operation of the BN series, successful operation of BN-600 during more than 30 years (lifetime was extended up to 45 years), allow to design and create the newest versions of this reactor type. Sodium brings specific problems: its chemical activity (burns in air), high neutron activation, positive reactivity effect, these problems however have been successfully dealt with during the mentioned period. The facility has three loops that increases its safety but makes it more expensive.

A lead-cooled reactor lacks these problems and possesses some advantages: low pressure in the primary circuit, natural coolant convection, high lead boiling point (1749°C), low scattering cross-section that allows achieving hard neutron spectrum, however has some disadvantages. The main problem is the high melting temperature of lead, 327°C, that defines the inlet and outlet temperatures of 400/500°C, thus requiring to maintain the reactor at hot conditions, which means high expenses to heat up and maintain lead in liquid state. Oxygen concentration in the circuit must be kept in a narrow region, to create a uniform oxide layer, which thinning can result in mass transport and lead-induced corrosion of constructions.

A Pb-containing reactor facility has an issue with radiohazard, since the radio-active polonium can result from bismuth that is formed by neutron activation of Pb (Pb^{208} (n, γ) $Pb^{209} \rightarrow$ Bi) [6]. Polonium is dangerous when released into gaseous environment and at coolant leakage. The coolant is toxic due to α- and β decay with half-life time of 10^6 years, and this makes problematic utilization of Pb, which amounts only in BREST-OD-300 up to about nine thousand tons, and makes hardly possible conservation of natural radiological balance.

The main problem is complexity of facility maintenance. This was one of the major reasons to decommission in 1990 s all Pb–Bi propulsionreactor facilities after only several core cycles.

Only nitride fuel is foreseen for the BREST reactor, since oxide fuel would require rigid fixing of FA to prevent floating up, while the oxide fuel together with pin clad is less dense than lead.

More dense nitride fuel is not enough studied yet. Experiments on several fuel pins only were conducted that have shown that the targeted fuel burnup cannot be reached, since PuN is swelled in much more extent than UN thus resulting in non-uniform clad mechanical load (swelling ~ 1,2% per 1% burnup [7]).

Economic problems

Economy is the final measure of competiveness. There is no economic assessment for the whole program nor for its parts yet. There is no such assessment even for BN-800, whose commissioning was scheduled for 2014. One can only make an approximate assessment, based on secondary sources.

If we consider specific metal intensity of reactor construction per KWel as the base parameter, this reads:

VVER	BN-600	BN-800	BN-1200	VVER-SKD
3,25 t/MW_{el}	13 t/MW_{el}	9,7 t/MW_{el}	5,6 t/MW_{el}	1,5 t/MW_{el}

The specific metal intensity corresponds to the relative cost of reactor facility construction. At present, VVER-1200 units are buildin Russia with the cost of ~ 3200 $/kW$_{el}$. This cost is already non-competitive with fossil fuel power plants. The BN-1200 is much more expensive compared to VVER-1200, the SVBR-100 reactor is assessed to 10,000 $/kW$_{el}$, and construction of BREST-OD-300 at the SKhK site will sum up to 64 milliards rubles (~ 7000 $/kW$_{el}$). These are approximate estimates that can grow several times during construction.

Also an on-site fuel cycle facility for radwaste reprocessing and new fuel fabrication is needed. It is assumed that it will add about 15% to the whole reactor cost [8]. And it should be noted that the MOX spent nuclear fuel will cost about five times more the initial fuel.

Minor actinides-related problems

One of the major problems of the spent nuclear fuel reprocessing is the presence of MA Am241—243 and Cm241–246, while these are the most intensive among the long-lived sources of radiation.

It is stated in the description of BN and BREST reactors that MA will be recycled with fuel – homogeneous burnup [8]. If MA is kept in the whole fuel, the latter is difficult to handle. A fully automated fabrication is than needed, new technologies of cooling down at each stage of fuel, pin and FA production. All this leads to considerable cost increase of the plant construction and operation.

As applied to the BREST reactor with nitride fuel, 1% of MA homogeneous content in fuel will considerably decrease the value of β_{eff} (β_{eff} ~0,35% without MA and 0,30% with MA [9]), that makes reactor reactivity control more difficult during prompt reactivity accidents or at loss of criticality.

VVER Spent fuel reprocessing

After the first core loadings with the fuel from the on-site fuel cycle facilities or fabricated at RT-1 of the Mayak production association and RT-2 – Mining and Chemical Combine (GKhK), Zheleznogorsk, the fuel cycle for fast reactors becomes closed at the breeding ratio ~ 1,2–1,3 with MOX (spent nuclear fuel) or when switched to the nitride fuel (BR ~ 1,05).

The VVER-440 fuel is reprocessed on RT-1 at the rate 400 t/year, and the VVER-1000 domestic fuel and from abroad, is stored in a repository. In GKhK, the RT-2 shall be commissioned in 2025 with capacity ~ 700–1200 t/year, herewith extracting ~ 10 $t_{(U+Pu)}$/year. In 2001, about 3000 tons of spent nuclear fuel has been accumulated in GKhK, and about 300 t yearly of domestic spent fuel as well as up to 20,000 tons from abroad [10] is planned, rising the need in additional storage and increased reprocessing capacity.

Fast reactors do not need plutonium – they can even have excess. Thus the question arises what to do with plutonium derived during the spent fuel reprocessing? It cannot be stored in a repository,the use of MOX (spent fuel) is not planned in VVER, moreover that in the considered program VVERs are to be replaced with BN or lead-cooled reactors.

Thus the conclusion is that spent fuel reprocessing is not needed and its repository is expensive. For a completely closed fuel cycle reactors with breeding ratio <1 are needed to burn all generated plutonium.

Development of the supercritical water-cooled reactors concept

Among six directions of perspective reactors mentioned in "Generation –IV", the most developed (after BN) are reactors with supercritical water. These reactors use proven technologies – BWR in the reactor core domain and TPP (thermal power plants) in the turbine domain.

Currently, the SCWR concept is developed in more than 15 countries (Japan, Korea, Canada, European Union, China and others).

Table 1 compares different SCWR designs: all reactors are single-circuit, with thermal neutron spectra (columns 1—5) and fast spectra (columns 6 and 7); fuel is oxide MOX.

Japanese projects with the thermal and fast spectra, financed since 2000 s, were taken as a starting point design; they are characterized by the unidirectional bottom-up coolant flow [11].

Later these two directions were developed and improved in many studies. In Canada, a vertical CANDU-SCWR is developed, with initially small power Ne = 300 MW, 120 fuel channels with 350°C at inlet and 625°C at outlet at the pressure of 25 MPa, followed by a commercial reactor of Ne = 1220 MW and thermal efficiency of ~ 50% [12].

Supercritical water reactor CSR-1000

Most rapidly the SCWR technology is developed in China. These works are included in national programs, financed, conducted by about a ten research centers and universities. At the ISSCWR symposium Chinese researches have presented a half (of 100) contributions devoted to SCWR reactors. They participate also in many European and Canadian projects.

The CSR-1000 reactor is characterized by thermal neutron spectrum to be launched in 2022 [13]. The first stage has been completed in 2012 – "development of CSR technology", approved by a government decision. This stage included the conceptual and engineering design of CSR power systems, experiments on heat exchange, thermal hydraulics and construction materials. Technical analysis of safety and accident operation are completed. In 2015 it is planned to obtain the government approval of the second stage for the period 2015–2017, where technical analysis for startup, shutdown, operational systems control, technology of tube fabrication for fuel pin cladding, perform irradiation experiments for the in-vessel construction materials and cladding materials, and irradiation of fuel pins in the core. The subsequent stages will be conducted until the construction and commissioning in 2022.

The Chinese researchers develop also a fast spectrum reactor [14]. The Russian project is taken as a basis. The Chinese reactor differs from VVER-SKD only in fuel pin diameter (they have 9,62 mm, while we have 10,7 mm) which resulted in a lower reactor power, Ne = 1530 MW (we have 1700 MW), the rest is taken without changes.

VVER-SKD reactor

Russia entered the GIF program in 2011. Starting 2006, as a result of joint work of OKB "GIDROPRESS", SSC RF-IPPE and "Kurchatov Institute", an advanced design of VVER-SKD with fast-resonance neutron spectrum, power Ne=1700 MW and bidirectional cooling scheme. This concept is considered as the major candidate of VVER technology improvement, including transition to U–Pu–Th MOX fuel and closing the fuel cycle [15–17].

According to the coolant scheme, the core is divided radially to the central and peripheral zones with approximately equal number of FA, 121 in the central zone and 120 in the peripheral one (Fig. 1). The peripheral zone is cooled by the top-down

Table 1
Characteristics of SCWR projects.

Parameter	SCWR (Korea)	SCLWR (Japan)	CANDU (Canada)	HPLWR (Europe)	CSR-1000 (China)	SCFR (Japan)	VVER-SKD (Russia)
Power, MW thermal electric	3989 1739	2273 950	2540 1220	2188 1000	2300 1000	3832 1698	3830 1700
Thermal efficiency, %	43,7	42	48	44	43	44,3	43,5
Temperature, °C water steam	350 510	280 508	350 625	280 508	280 500	280 523	290 540
Steam pressure, MPa	25	25	25	25	25	25	25
Water mass flowrate, kg/s	2518	1816	1312	1113	1190	1897	1880

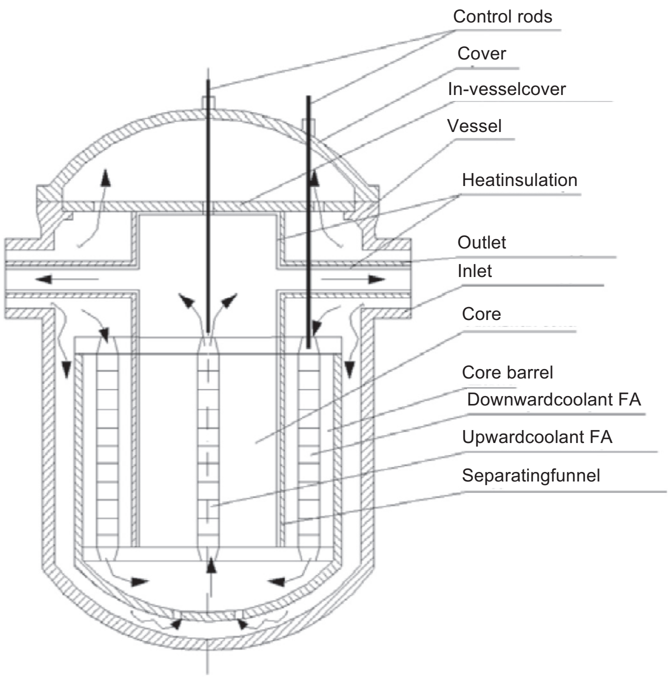

Fig. 1. Reactor coolant scheme.

coolant flow. At the core bottom, the coolant exiting peripheral FA is collected in the mixing chamber and passed to the central zone inlet that is cooled by bottom-top coolant flow. Coolant flows in the downward and upward legs is separated at ~ 385°C. In the downward leg the coolant is heated up to 95°C, its density is changed approximately three times. In the upward leg, the coolant heating amounts to 155°C, the density changes 2,2 times. Thus, the neutron spectrum vertical change is small, but changes radially, and no fuel enrichment profiling is necessary for heat deposition distribution flattening, the void effect is negative in the absence of blanket, all FA constructions will operate at the temperature gradient twice smaller, as compared to the unidirectional scheme [11]).

VVER-SKD expected advantages

- Fast-resonance neutron spectrum admits high fuel breeding ratio (close to 1), reduce uranium consumption, ensure utilization of U238 and radioactive waste burnup.
- Thermal efficiency is increased up to 44–45%, compared to 33–34% of existing NPPs.
- Decrease of the core coolant mass flowrate due to possibility to increase the coolant heatup by 250°C, as compared to 30–35°C in VVER, and consequent reduction of pipe diameters.
- The once through coolant system makes unnecessary steam generators and all equipment of the secondary coolant circuit.
- Use of in-series produced turbine hall equipment, widely applied in fossil power engineering (turbines, heaters etc).

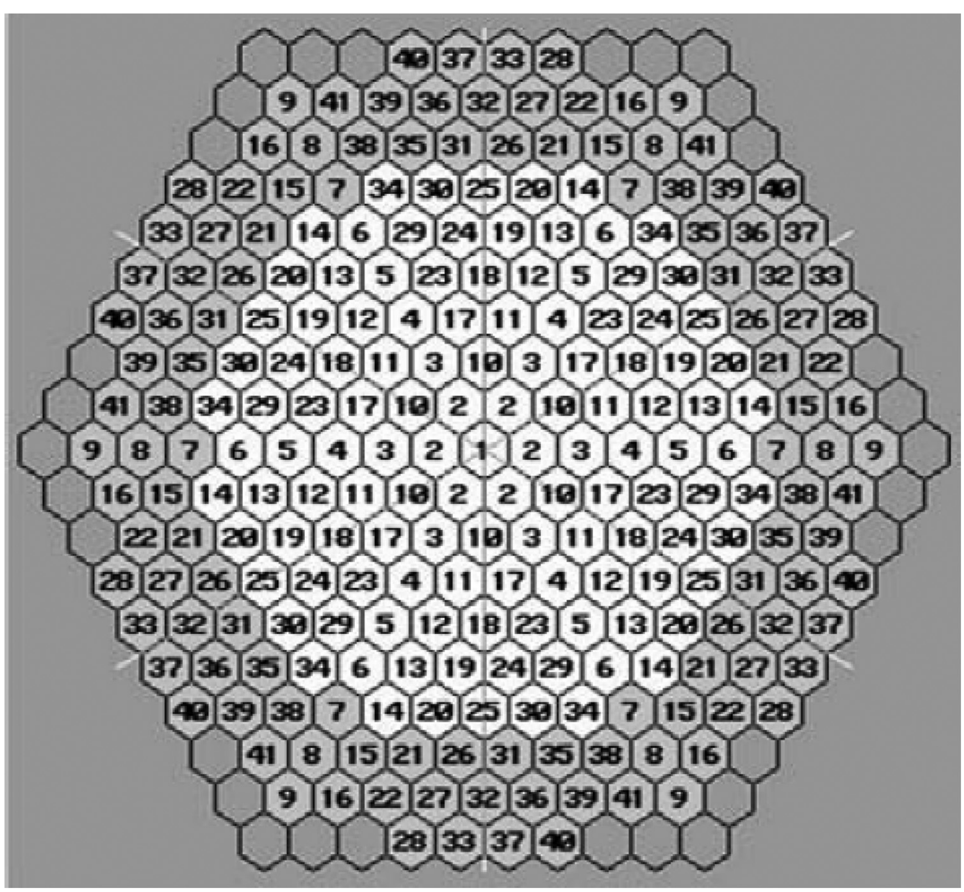

Fig. 2. Core loading.

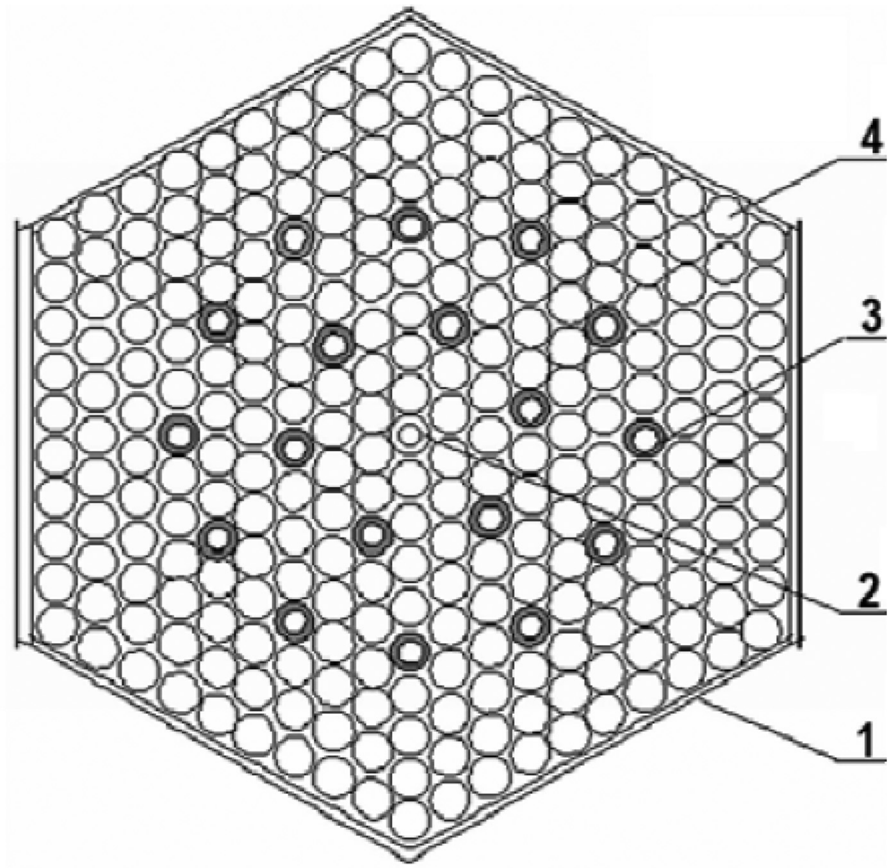

Fig. 3. FA horizontal cross-section: 1 – wrapper with 2,25 mm wall thickness; 2 – central tube 12,0 mm x 0,55 mm; 3 – 18 absorber rod guide tubes, 12,0 mm x 0,55 mm; 4 – 252 fuel pins with clad 10,7 mm x 0,55 mm and 12 mm pitch. Construction material – stainless steel EP-172 (ChS-68).

- Considerable decrease of the containment volume and construction amount, the specific metal intensity will be ∼ 1,5 t/MW_{el}.
- Reduce of the operational cost.

Vver-Skd reactor in closed fuel cycle

Fuel cycle neutronics calculations

Figs. 2 and 3 show the calculation model core loading and the FA horizontal cross-section corresponding to the bidirectional scheme of coolant flow with peripheral and central core zones.

Fuel material is a mixture of VVER spent fuel and weapon plutonium.

At the effective density of the uranium and plutonium oxides of γMOX = 9,5 g/cm3, the weapon plutonium oxide density accounts to 0,7 g/cm3 and is equal in all FA.

As an alternative to MOX (U–Pu) fuel, the use of Th in mixed core loadings has been considered: U–Pu in the central core zone and U–Th – in the peripheral. The nitride fuel has been also considered, with the density 80% of theoretical γ(U,Pu)=11,5 g/cm3 at 9% of weapon Pu content. Results of computational modelling of fuel cycles are presented in Table 2.

From the presented results one can see that due to reactor properties (fast-resonance neutron spectrum, bidirectional coolant flow scheme with more dense coolant in the peripheral zone) the core voiding does not represent a problem (the void reactivity effect is negative during the whole core cycle). To ensure required reactivity worth of compensating control rods in the most demanding reactor state – flooded with cold water, enriched boron is needed in absorber rods, but even in this case the variant with U233-Th core loading requires insertion of gadolinium.

The nitride loaded core possesses reactivity excess of $\Delta K \sim$ 0,44% at BoC and 0,08% at EoC, thus reactor safety is ensured at withdrawal of all control rods.

Investigation of MA burning capabilities of VVER-SKD

The VVER-SKD can be effective in the closed fuel cycle, since it uses its own spent fuel with small addition of plutonium (160–200 kg of weapon or reactor-grade). The (U–Pu–Th)-based fuel cycles can be used as well.

Handling of MA becomes the major problem, mainly Am241–243 and Cm242—245 that define most radioactivity of spent fuel and radioactive waste. The Np237 isotope is not separated from fuel, curium due to its high heat deposition is better to separate to a long-term repository, where it decays into plutonium.

Fig. 4 shows horizontal cross-section of the FA containing minor actinides.

Table 2
Reactor main characteristics for U–Pu–Th fuel cycles.

Parameter	U–Pu	Pu–Th	Th	(U, Pu)N
Initial core loading, *t*	135,6	137,3	139,0	167,9
Amount of fissionable isotopes at initial core loading, Pu/U233, *t*	11,7/0,0	5,914/4,80	0/10.81	13,39/0
Fissionable isotopes per FA, Pu/U233, kg	48,86/0	48,86/39,99	50,24/39,46	55,57/0
Fuel enrichment, Pu/U233, % Central zone Peripheral zone	7,7/0	7,70/0	0/9,0 0/	7,88/0
	7,7/0	0/7,0	6,9	7,88/0
Number of core reshuffling	5	5	5	5
Core cycle length, effective days	300	310	300	295
Fuel burnup, average/maximal, MW day/kg t	39,79/65,4	42,2/68,6	34,6/47,5	30,7/53,5
Power peaking factors, Kq/Kv	1,46/2,19	1,61/2,62	1,67/2,8	1,71/2,66
Fissionable isotopes loading, t/year	2,34	2,11	2,20	2,62
Fissionable isotopes discharge, t/year	2,18	1,87	1,96	2,582
Breeding ratio				
Central zone	1,013	1,003	0,957	1,059
Peripheral zone	0.853	0.769	0,800	0,909
Average	0,933	0,887	0,890	0,984
Void effect, dK% BoC/EoC	−5,88/−3,64	−3,24/−1,40	−6,28 /−2,32	−0,005/−0,003
KeffBoC/EoC	1,0175/1,0010	1,0281/1,0010	1,0344/1,0000	1,004/1,001

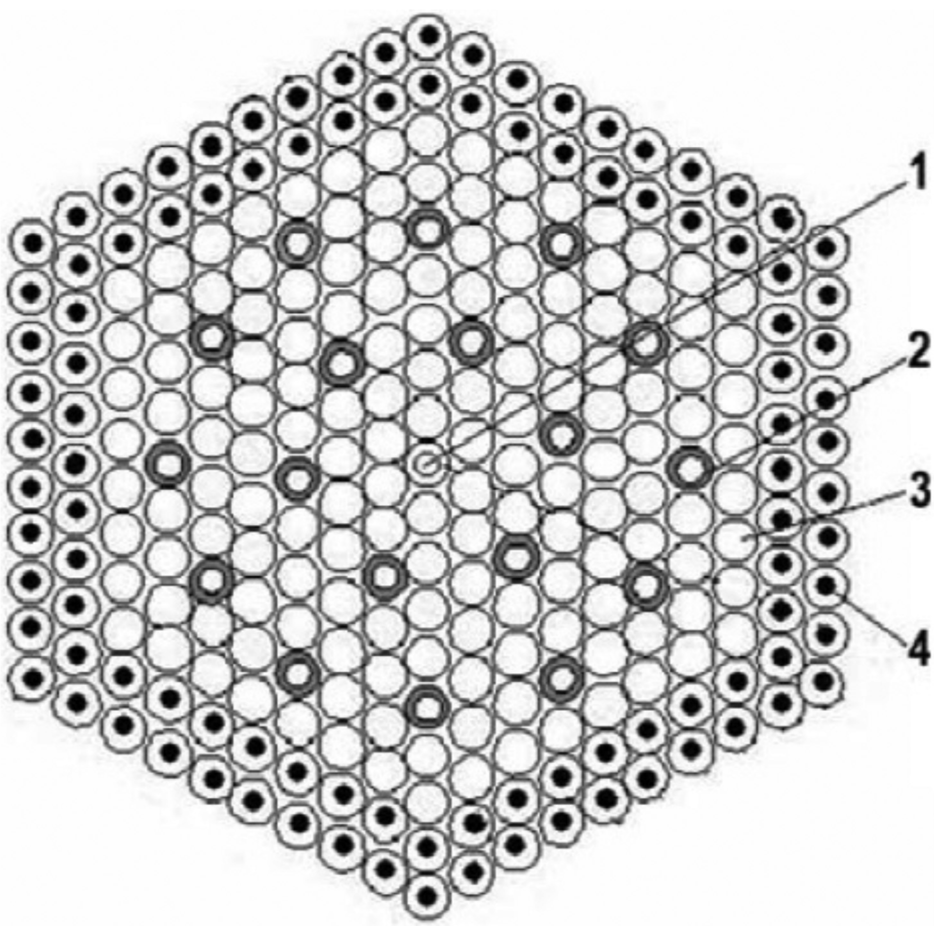

Fig. 4. FA horizontal cross-section: 1 – central tube; 2 – control rod guide tubes; 3 – MOX fuel pins; 4 – fuel pins containing MA.

A deeper burnup can be reached in VVER-SKD when the MA containing FA is loaded to the peripheral zone during two core cycles [18]. In 10 years operation, VVER-SKD accumulates ∼1400 kg of MA (97% Am and 3% Cm) in equilibrium cycles with 5 year cycle for standard FA and 10 year cycle for MA containing FA.

Fuel cycle calculations were made for hexagonal geometry and 5 groups approximation, with the help of the WIMS-ACADEM code.

Fig. 5 shows 60° symmetry core part with results of fuel burnup MW d/kg U and FA power peaking factors Kq, for the loading with 4 FA in the peripheral zone (# 8, 23, 36, 45)

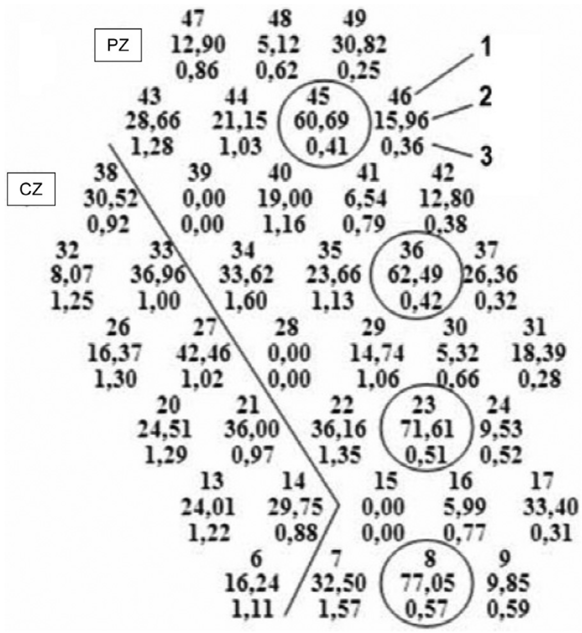

Fig. 5. Fuel burnup and FA power peaking factors in the core 60° symmetry part at EoL: 1 – FA number, 2 – burnup E, MW d/kg U; 3 – Kq; MA containing FAs are denoted by cycles; PZ – peripheral zone; CZ – central zone.

containing MA, at the EoL after 10 years of equilibrium core cycles. FAs # 15, 28 39 represent the separating funnel and correspond to its volume and composition.

MA content in MA containing FAs decreases to 12% of initial in 10 years of irradiation, and these fuel pins can be shipped to long-term repository. 24 FAs are loaded with 1230 kg of Am241–243. Thus, all MA generated during 10 years of reactor operation can be burnt in this time, about 24 FA are needed for this.

MA containing FAs consist of the mixture 35% ZrO and 65% AMO2. This composition is not soluble in acids and water during long-term repository.

MCNP calculations showed that the maximal over cycle peaking factor does not exceed 1,3.

The use of MA containing FAs (in specified amount) saves about 40 kg of plutonium per year, without deteriorating FA power peaking factor in the core.

Conclusion

Foreign VVER-SKD research review reveals their systematics and consistency, project cooperation and, most important personnel advanced training. Foreign institutions, conduct together with IAEA courses targeted to SCWR design and technology. Demonstration facilities construction are planned to 2022, and commercial – to 2030 and they will replace existing water-cooled reactors of generation 3 and3+.

Despite the fact that Russia has joined the SCW-related work within the framework of GIF, no project agreement about participation in particular international projects was signed so that we cannot use the results achieved.

As it was shown, the CFC based on merely BN and BREST reactors has lot of technical and financial difficulties, not all CFC-related problems could be solved and the VVER-CKD can solve them in many respects. Introduction of these reactors could be improve economy, reach deeper MA burnup and considerably reduce the amount of nuclear hazardous work by using reprocessed spent nuclear fuel (U+Pu).

The expertise accumulated during the last decade helps to elaborate in details earlier developed concept, identify plans of priority research, compose technical task description and start the detailed design of a small power experimental reactor, ~30 MWt. Development of such a reactor characterized by universal cooling schemes and neutron spectra can be organized on international level.

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